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Abstract

KINETICS OF THE HOMOGENEOUS AND HETEROGENEOUS OXIDATION OF URANIUM (IV) AND ITS ACID LEACHING PROCESSES

E.A. Kanevsky, I.A. Yakubovich et al.

The comparison of the standard potentials of oxidation reduction processes with the uranium (IV) oxidizing rate by the different oxidizers in the acid solution shows that kinetic factors are the most important. In the absence of oxidizers the dissolving UO_2 cannot be practically observed and when the oxidizer is present the dissolving rate coincides with the oxidizing one.

The kinetics of the oxidation of uranium dioxide by manganous dioxide in the sulfuric acid solution is mainly determined by the conditions of the contact of the solid phases. In the case of the introduction Fe(III) ions they function as the direct oxidizers in spite of lower oxidation-reduction potential of this system.

The investigation of the process of oxidation of UO_2 by Fe(III) ions shows the high effectivity of hydrolyzed ions $FeOY^{2+}$ and $Fe(OH)_2^+$.

It has been established that the reaction is of first order with respect to Fe(III) and the disproportioning rate of U(V) in the sulfuric acid solution is higher than the oxidation rate UO_2 by hydrolyzed ions Fe (III). Among these ions $Fe(OH)_2^+$ is the most active.

In the presence HNO_3 the kinetics of the oxidation and the dissolving of UO_2 is determined by the rate of destruction of HNO_3 and the presence of the Fe(III) ions does not have such importance as in the system UO_2-MnO_2-Fe (III). At the low temperature Fe(III) ions do not accelerate the process. The causes of this effect have been discussed.

The investigation of kinetics of the oxidation of U(IV) in solution by the chlorites shows that in this reaction the active part of oxidizer are

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the products of destruction of chlorites. The rate of the interaction UO_2 with the chlorates in the acid solution is determined by the concentration of undissociated chloric acid.

The explanation of the slow proceeding of the reaction at the low temperature is given. The catalytic influence of the iron and vanadium ions on this process has been investigated.

The importance of these conclusions for the understanding of kinetics of the acid leaching of uranium from the ores has been discussed.

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Abstract

CHOICE OF SOME OPTIMUM CHARACTERISTICS OF CONTROL ELEMENTS FOR NUCLEAR REACTORS

V.V. Orlov, V.S. Andreyanov, E.I. Grishanin et al.

The first part of this paper describes a method of determination of optimum burnable absorber characteristics and the technique of distributing them over the core to flatten the shape of heat release during the reactor life-time.

Requirements for the absorption cross-section of burnable isotopes and some techniques of using burnable absorbers are formulated. A possibility of reaching zero unbalance of reactivity is discussed based on an ideal case of infinite absorption cross-section. The practical recommendations for choosing the geometry of rods are given on the basis of this analysis. As specific examples the following practically important types of burnable absorbers are investigated:

(1) homogeneous,
(2) non-large self-shielding factor,
(3) large macroscopic absorption cross-section.

The second part of this report deals with the method of determining optimum composition of an arbitrary mixture of resonant absorbers for control rods from the standpoint of increasing rod efficiency, decreasing distortions of neutron fields due to the rods etc.

The method is illustrated by an example of optimum mixture corresponding to the maximum efficiency of rods of given dimensions in intermediate reactor $\Pi\Phi - 4$ $/\frac{\rho_{Bs}}{\rho_U} = 90/$. Efficiency increases by 30% (compared with those of the same sized Ir rod) were observed for optimum mixture of Ir, Re, Eu, Dy, Hf.

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This method has also been used to calculate the optimum plate control rod composition for water-moderated water-cooled reactor/$\frac{\rho_H}{\rho_u} = 280$/ , corresponding to minimum neutron field distortions near these rods.

Calculated results are compared with experimental data.

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Abstract

OPTIMIZING THE CHARACTERISTICS OF POWER REACTORS BY MEANS OF NON-UNIFORM DISTRIBUTION OF MATERIALS

E.S. Gloushkov, E.I. Inyutin, et al

The paper deals with the problem of optimizing various parameters of power reactors (energy release profiles, temperature fields, etc.) by means of non-uniform distribution of materials within the reactor.

The problem under discussion is given, theoretical substantiation and methods are described thereby the reactor systems can be calculated so as to obtain optimal parameters of the reactor. The paper also describes the relevant computer code.

The paper dicusses the experimental techniques and the results of experiments in varying the energy release profiles by means of non-uniform distribution of fissionable material and moderator within the core. The experiments were carried out on critical assemblies with ligt-water, organic and beryllia moderators and consisted in measuring the flux and fission integral distribution for various laws of distribution of fissionable material and moderator. The experimental results are compared with the theoretical characteristics of systems.

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Abstract

NEUTRON MODERATION IN HYDROGENOUS MEDIA

L.N. Yurova, A.A. Polyakov, et al.

This report deals with neutron moderation and diffusion in various hydrogenous media.

The first part of the report describes the method of measuring neutron spatial distribution in "bounded" media and presents some results of experimental investigation of spatial distribution of neutrons in various media.

The effect of the dimensions (width) of the neutron fission source and of the geometrical structure of the medium (heterogeneity) upon the neutron spatial distribution was determined for those neutrons which have been moderated down to 1.46 ev.

The second part of the report presents the results of the thermal neutron diffusion measurements performed by pulsed technique for a number of hydrogenous substances in a temperature range from t melting to t boiling.

The diffusion coefficient was obtained in the form

$$D(v) = \text{const.}\ v^{a},$$

where a is the function of molecular structure of the substance. The value D(v) allows one to calculate the transport length and transport cross-section per atom of bounded hydrogen.

It was found that the value of the transport cross-section per atom of hydrogen strongly depends upon the molecular structure of the investigated hydrogenous medium. The example for $[C_{15}H_{16}]$ was a = 2.58 and σ^{H}_{t2} = 86 barns, for $[C_{12}H_{10}]$ a = 1.33 and σ^{H}_{t2} = 31 barns.

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Abstract

AN INFLUENCE OF CROSS-SECTION RESONANT STRUCTURE ON NEUTRON DIFFUSION AND MODERATION AND RESONANCE EFFECTS FOR FISSIONABLE NUCLEI

L.P. Agabyan, I.I. Bondarenko and A.G. Gordeev

The way of taking into account the resonant structure of neutron cross-section in the reactor theory is considered. These effects are shown to be of considerable and sometimes of major importance for the determination of neutron space energy distributions even for the fast reactors.

The method is developed for taking into account the resonant structure in multigroup reactor calculation for Pn- and Sn-approximations.

The theory for evaluation of resonant structure parameters is described for different energy ranges. For isolated levels this theory is rigorous but only qualitative estimates of the effects in question can be made for partially overlapping resonances.

It is shown also that because of the interference between potential and resonant scattering the cross-section fluctuations in the region of overlapping resonances are strong enough for resonant self-shielding to be quite significant in this energy range.

Theoretical conclusions need experimental proof. In this report some methods are described for measuring parameters of resonant structure of cross-sections which affect the neutron diffusion in media. These methods do not require high energy resolution and can be applied at any neutron energy including the region of overlapping levels.

The experimental results obtained by these methods are summarized and compared with the theory.

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The mean values and statistical distributions of fission width are important characteristics of the resonant structure. Some theoretical conceptions on fission-channel scheme were used to make reasonable choice of these characteristics. The Bohr-Wheeler formula for average fission-widths was used to analyze experimental data for reactions (d,pf), (n,f) and (γ,f). These data are found to be in agreement with the conception of fission channels proposed by O. Bohr, negative parity channels being by 0.6–0.8 MeV above the first positive parity channel. In this case the Bohr-Wheeler formula is shown to describe quantitatively the fission widths of neutron resonance of U^{233}, U^{235} and Pu^{242}. One must assume Pu^{239} having the ground state of negative parity to explain the sharp disagreement between theory and experiment for this nucleus.

Experimental distributions of fission and reduced neutron widths of neutron resonances were analyzed. The existence of two sub-systems of resonances with different spins and possible omission of weak levels were taken into account. Resulting numbers of fission channels are N = 1 for Pu^{239} and N $\geq$ 3 for U^{233}.

The dependence of $\bar{\alpha} = \frac{\bar{\sigma}\gamma}{\sigma}$ on energy for s- neutrons was calculated ($\alpha = \frac{\bar{\Gamma}\gamma}{F_t}$ was taken to be constant). $\bar{\alpha} > \alpha$ always the difference between two values being especially large at neutron energy En$\sim$100 eV. With an increase in neutron energy when the contribution of neutron width increases and fluctuations of the total width decrease, the value of $\bar{\alpha}$ approaches α, i.e. diminishes even before neutron p-wave becomes significant. The maximum effect of this kind is expected for Pu^{239} - the nucleus with the most broad distribution of fission widths and with greatest reduced neutron width.

These results were also used for calculating temperature effect for Pu^{239}.

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Abstract

NEUTRON-PHYSICAL PARAMETERS OF URANIUM-WATER LATTICES

L.G. Andreev, I.Kh. Ganev, S.I. Garanin et al.

The paper deals with the study of U-H_2O lattices having rod-type fuel elements. It presents the results of three independent experiments carried out at the Kurchatov Atomic Energy Institute and certain problems involved in calculation of parameters of uranium-water systems.

1. Critical experiments with lattices at 7, 8 and 14 mm spacings, with and without absorbers, are described. The core consisted of fuel elements similar to those used in the reactors of the icebreaker "LENIN" (diameter of sintered uranium dioxide fuel element - 4.5 mm, enrichment - 5.3%, cladding - 0.3 mm stainless steel or 0.75 mm zirconium). The absorbers were made of a mixture of boron carbide and aluminium oxide in the form of rods from 4.4 mm to 12 mm in diameter. Their "blackness" dE_c varied within the range of 0.75 to 40.

The paper describes the one-group method developed for calculating cores having complex configuration, with corrections for absorption of neutrons in the epithermal region of the spectrum. It is shown that the theoretical and experimental data agree within 10%.

2. An investigation was performed of the parameters of lattices with spacings 13, 15, 17 and 19 mm and with the elements made of sintered dioxide of natural uranium (fuel element diameter - 9.4 mm, cladding - 0.7 mm aluminium). Neutron absorption ratios were measured for various components of the core, as were the fission cross-section ratios of Pu^{239} and U^{235}. The paper presents the values of the multiplication factor K_∞, its four co-factors μ, φ, θ, ν eff. and the breeding ratio ρ with the neutron leakage taken into account.

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For the 13 mm spaced lattice the value of the material buckling $\varkappa^2$ was determined from the spatial distribution of neutrons. The full migration area M^2 was measured by the poisoning method as were the migration areas at moderations down to the fission threshold of U^{238} - M_8^{+2} and down to the main resonances of U^{238} - M_8^{22}. The form of the critical equation was estimated on the basis of the experimental data and was shown to be close to a one-group equation. Parameters K_∞, μ, ρ, etc. were likewise determined for a dry uranium lattice.

3. A 60 mm spaced lattice with uranium-magnesium ceramic (fuel element diameter - 35 mm, magnesium weight content - 17.8%, enrichment - 2%, cladding - 1 mm aluminium) was subjected to a complex investigation. Using a method of pulsed source the dependence of the neutron flux relaxation time on the height and radius of the assembly was studied in a wide range of the geometrical bucklings (from critical value $\varkappa_o^2$ to $3\varkappa_o^2$). Critical configurations and the corresponding neutron flux distributions were determined. Provision was made for the water reflector to be separated from the assembly by a cadmium sheet.

Parts of the function describing variations of the flux in time and space were determined theoretically, this was important for carrying out the experiment. It was shown that the measurements in uranium-water systems can be analysed using a theory which disregards the correlations between the place and the moment of neutron absorption. The corrections for these correlations and for the heterogeneity and deviations of the capture and fission cross-sections from the law 1/V are insignificant.

The anisotropy of the migration area M^2 was detected and the next space moment $(r_4)_{aver.}$ was estimated. Parameters K_∞, M_2^2, M_H^2, $\varkappa_o^2$, life time, radial and axial savings, etc., were also measured.

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Abstract

EXPERIMENTAL INVESTIGATION ON PHYSICAL CHARACTERISTICS OF WATER-WATER REACTORS FOR SMALL POWER PLANTS

I.G.Morozov, E.I.Iniutin, M.N.Lantsov,
E.A.Plaksin et al

This paper presents the experimental results on physical characteristics of water moderated and cooled reactors with uranium fuel enriched in U-235 up to 10-20 per cent. The results were obtained since 1959 through 1963. Such reactors are useful im small atomic power plants (including mobile plants). The analysis of experimental results, as a rule, is made on multigroup theory basis.

Critical assemblies (cold and hot) on which the experiments were performed are discribed in the report.

The main presented results are the following.

For more accurate determination of design values of fuel charge in cores and reactivity excess a number of critical assemblies were constructed. These assemblies differed in core content (types of fuel elements, fuel enrichment, lattice pitch, etc) and its configuration.

Excess reactivity determination of a full-sized core was carried out by integration of experimental values of $d\rho/dh$ against the core height. In these experiments core heigh variation was accomplished by a varaible charge of a heterogeneous absorber, by boron contaminating og the moderator, as well as by core radius variation. The common method of $d\rho/dh$ measurement was modified according to the peculiarities of the systems with a great number of channels with small

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passage section.

The great excess reactivity inherent to the reactors considered and the peculiarities of temperature effect of their cores compell severe requirements to the reactivity compensation organs. The paper contains the experimental results on determination of compensation effectiveness of rods with varaible construction and content (Eu,Ag, B,Cd) of compensating lattice and burning-out additives. In addition, results on determination of perturbation of heat release field by core lattice regularity disturbances (including the case of absorption change of controls).

The emphasis is made on the experimental investigation of interference of controls. The distribution of compensating unit functions according to their use for physical profiling water-water reactor cores is considered.

Considerable reactivity changes with temperature are inherent to water moderated and cooled reactors. Experimental results on studies of warm-up curves for such reactor cores within the temperature range from 20°up to 200°- 300°C are given in the paper. The warm-up of undisturbed (with regular lattice without compensating elements) cores as well as of critical assemblies with disturbed lattices practically accomplished in water reactors was carried out. Thermal neutron field variation across the lattice cell with temparature was investigated experimentally. Experimental data on control compensating effectiveness variation with core temperature are given in the report.

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Abstract

EFFECT OF CHEMICAL BINDING ON NEUTRON THERMALIZATION

L.V.Maiorov, V.F.Turchin et al.

The paper discusses the effect of chemical binding on the integral and differential scattering characteristics and the formation of slow neutron spectra in the most important moderators (C, H_2O, Be, ZrH, LiH).

The differential scattering cross-sections which are calculated with the aid of the UPRAS code taking account of the acoustic, optical and diffusive motions of the atoms, are used for the analysis of the neutron spectra in heterogeneous media, media with a temperature gradient and media with a pulsed source.

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Abstract

NEUTRON-PHYSICAL CHARACTERISTICS OF ZIRCONIUM HYDRIDE-MODERATED SYSTEMS

N.N.Ponomaryov-Stepnoy,
O.N. Smirnov, V.G.Kosovskiy, et al

The work contains the results of experimental and theoretical research of physical clusters with zirconium hydride moderator and highly enriched uranium.

The studied clusters are a system of alternating layers of fuel and moderator. The flat-type fuel elements consist of teflon-4 mixed with uranous-uranic oxide of 90% enrichment. The moderator blocks made up of zirconium hydride ZrHn with various content of hydrogen are shaped and sized so as to ensure arrangement of moderator in layers of different thickness.

The required information on the properties of such reactor system was obtained by carrying out critical experiments and by taking measurements on critical clusters.

The results of the critical experiments with unreflected systems ZrHn + U^{235} have been generalized and presented in the form of diagram $\mathcal{L}$ - C ($\mathcal{L}$ -material buckling, C - con-

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centration).

The effect of fissionable substance self-shielding on the reactor critical parameters has been studied.

Efficiency of various reflectors (Be, BeO, ZrHn) has been analized.

Experiments are described devoted to studying the effect of various absorbers uniformly distributed in the core volume on the critical parameters.

The neutron spectral characteristics in the tested clusters have been measured. The results of measurements taken for determination of space distribution of neutrons in reactor volume and in core cell are cited.

The proper method for carrying out neutron-physical calculations of such quasi-homogenous reactor systems with zirconium hydride moderator has been selected.

Comparison is drawn between the calculated and experimental data.

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Abstract

NEUTRON-PHYSICAL CHARACTERISTICS OF REACTOR SYSTEMS WITH BERYLLIA AND BERYLLIUM MODERATORS

V.P.Garin, E.S.Glushkov, R.K.Goncharov et al

The paper describes the results of the experimental study of the neutron-physical characteristics of systems with Be and BeO moderators. The results of critical experiments are given for systems of the following two types:

1. Systems close to homogeneous ones with highly enriched uranium.

2. Heterogeneous lattices using natural and slightly enriched uranium.

For systems of the first type the experiments included the investigation of the critical parameters, neutron spectrum, effect of the reflectors, etc, at widely varying concentrations of the fissionable material and geometrical characteristics of the assemblies (from fast to thermal spectra); investigation of the dynamic characteristics of the systems; investigation of the effectiveness of control components made of various materials depending on their position in the critical assemblies.

The multi-group methods of calculation of such assemblies were corrected on the basis of the experimental data.

For systems of the second type the experiments included the investigation of the critical parameters of the lattices (K_∞ and its co-factors), space-energy distribution of the neutrons in the cell with the aid of a mechanical selector. Contribution to K_∞ made by the reactions $(n, 2n)$ and (γ, n) on beryllium is assessed. In the case of pure moderators the paper gives the results of the experiments in finding neutron age up to various energies, square diffusion length diffusion constant, etc.

The impulse method of studying the system characteristics was used in the experiments among other methods.

The results of calculations showing the importance of (n,2 n)process in neutron multiplication in the reactor are given in the paper.

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Abstract

METHODS OF SOLUTION OF TRANSPORT EQUATION IN INHOMOGENEOUS AND FINITE MEDIA

S.M. Ermakoff, V.G. Zolotuchin, F.F. Michajlus et al.

The report consists of three parts. The first part is devoted to finite difference methods of solution of plane-parallel and spherical-symmetric γ-quanta and neutron transport problems in hydrogenous media. In the second part some methods for the improvement of the statistical efficiency of Monte-Carlo methods for the case of radiation transport problems in inhomogeneous and finite media are considered. The third part is devoted to the Monte-Carlo study of the neutron penetration through the resonance absorber slab.

1. To formulate a finite-difference method of solution of the integro-differential transport equation it is necessary to consider separately two problems: (a) a finite-difference representation of the differential part of the equation and the solution of the system obtained; (b) an approximate estimate of the scattering integral occurring in the equation.

In the case of plane-parallel problems the formulation of part (a) is not difficult. When the spherically-symmetric problems are investigated the method similar to the Carlson's S_n-method can be used. The difficulties due to singularities of the equation at the origin are avoided by means of the Vladimirov's characteristics method.

For the formulation of the part (b) both for the plane-parallel and spherically-symmetric problems the scattering integral is written down as a double integral along the surface of unit sphere and then evaluated using the simplest quadrature formulae so constructed that the main features of the scattering kernel and the relatively weak dependence of the directional distributions on the energy are taken into account.

In practice the energy interval is divided into sub-intervals and in each of them the spectral distribution is represented by a linear function of energy with

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coefficients depending on directional and spatial co-ordinates. At each fixed value of energy the integro-differential equation is considered to be a differential equation with a known right side. An appropriate choice of the quadrature formula for the approximate calculation of the scattering integral makes unnecessary the use of iterative procedures.

A range of applications of methods described can be extended appreciably by means of the adjoint function theory.

2. In the second part of the report problems of calculating differential angular and energy distributions of the radiation at a fixed point in space (local flux calculation method) are examined. This method proposed by Zerby and collaborators has an infinite standard deviation which considerably reduces its efficiency.

A modification of the local flux calculation method is proposed which will have a finite standard deviation for the total flux or dose rate at a point fixed in space.

Calculation of Fourier coefficients of the flux function in terms of space variables also ensures finiteness of the standard deviation.

In this case the slow convergence of Fourier series may be improved by the choice of a weighting function which takes into account the singularity of the integrand.

An application of the local flux calculation method to the problem of penetration of radiation over a small surface element is also discussed in the paper.

The mathematical aspects of the problem arising as a result of the application of random quadrature formulae of higher accuracy are discussed.

New quadrature formulae and evaluation of their errors are given.

Examples of the calculation for one-velocity problems with an isotropic scattering are given.

3. A transfer of neutrons in resonance absorber slab is considered. For this purpose the Monte-Carlo method using importance sampling of initial energies of neutrons proved to be very effective. A specially constructed function is used as an adjoint function.

This approach makes it possible to perform neutron absorption and penetration calculations for the wide range of sample thicknesses.

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Abstract

SOME NEW ASPECTS OF USING ADJOINT FUNCTIONS AND PERTURBATION THEORY IN REACTOR AND SHIELDING CALCULATIONS

L.N. Usachev, A.I. Nevinnitza, M.F. Troyanov et al.

Some aspects of applications of adjoint equations to investigate some reactor and radiation shielding problems are presented.

In this report a new application of perturbation theory recently developed by one of the authors (Atomnaya Energfya, No.12, 1963) is considered in which one can obtain not only reactivity changes but also variations of a number of other reactor characteristics represented as a ratio of any two values of neutron-induced processes in reactor. Among such values one can mention for example a breeding ratio. Computation algorithm is described and some applications of this perturbation theory are given.

In this report adjoint equations and perturbation theory technique are applied to find minima of reactor critical mass in general case. Conception of efficiency function of fission material in reactor is introduced. Constancy of this function within reactor core is shown to be the condition of reactor critical mass to be minimum. The algorithm of numerical calculations of minimum critical mass based on iterations with the efficiency function of fission material having a constant value is discussed. Material efficiency conception in reactor and corresponding algorithm for the determination of this optimal distribution of material can be extended to other reactor materials (for example, moderator, absorber, U^{238} distribution in breeder reactor).

In addition to the above-mentioned applications of adjoint equations to reactor problems the applications of non-homogeneous adjoint equations to investigation and design of radiation shielding are also considered. Physical meaning of the adjoint function as hazard function for neutrons with regard to processes in

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shielding is shown. The perturbation theory formulae for the most general case of medium and source parameter changes are derived.

Efficiency function conception of shielding material for various processes in shielding is introduced.

The condition of optimal shielding material distribution is formulated on the basis of mentioned efficiency function concertion.

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Abstract

CALCULATION OF SLOW NEUTRON SPECTRA

G.I.Marchuk, G.A.Ilyasova,
V.N.Morozov et al.

The paper reviews the methods of calculation of slow neutron spectra. A multi-group kinetic equation is given for the flux and value of neutrons. Methods were devised thereby the slow neutron spectra can be calculated using the spherical harmonic method, Carlson method and Monte-Carlo method. A comparative analysis is made of the calculations of the neutron spectra by different methods. Angular distributions of the neutron flux are given. The calculation results are compared with the experimental data obtained in the I.V. Kurchatov Atomic Energy Institute.

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Abstract

THE STUDY OF NEUTRON THERMALIZATION IN WATER AND GRAPHITE-WATER REACTOR LATTICES

E.Ya. Doilnitzin, A.G. Novikov, Yu.Yu. Glaskov et al.

The investigation is described of thermal neutron spectra for water and graphite-water reactors by means of the time-of-flight method.

Space and energy distribution of neutrons in water with a Fermi-spectrum source is presented.

Neutron spectra are measured in water poisoned by an absorber according to I/v law. The results are compared with the calculations for a gas model, taking into account the chemical binding forces.

The relaxation of a neutron gas temperature in water has been studied.

Two approaches are considered fairly explaning the asymptotic spectrum for thermal neutrons.

The results of measurement and those of calculations with P-3 approximation are given for several types of graphite-water reactor lattices, taking into account the thermalization effects.

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Abstract

EXPERIMENTAL WORK ON NEUTRON THERMALIZATION

V.I. Mostovoy, B.S. Dikarev, I.P. Eremeev et al.

The paper reviews the experimental work on neutron thermalization which has been carried out in recent years in the Kurchatov Atomic Energy Institute. This work deals with the most interesting aspects of neutron thermalization which have practical importance, and is carried out along three lines.

1. Investigation of inelastic scattering of slow monochromatic neutrons on bound nuclei of the moderator.

In connection with this line of research the review gives the results of double differential scattering cross-sections of slow monochromatic neutrons $\sigma(EE'\theta)$ in hydrogen-containing moderators. A pulsed flux of monochromatic neutrons of energy E incident on the sample was separated from the horizontal beam of thermal neutrons of the VVR-M reactor (Physics Institute of the Academy of Sciences of the Ukranian SSR) with the aid of a mechanical monochromator. The resolution of the monochromator was independent of its speed of rotation and was equal to $\frac{\Delta E}{E} = 0.2$. The energy E' of neutrons scattered from the sample at an angle θ was measured by the flight time method with a resolution of about 25 μsec/m.

The double differential scattering cross-sections were measured within the range of energies of the incident neutrons from 15 to 300 eV at scattering angles from 15° to 120°. Certain data were obtained concerning the effect of the sample temperature on the double differential scattering cross-section.

The comparative analysis of the cross-sections and average characteristics of neutron scattering in water and monoisopropyldiphenyl ($C_{15}H_{16}$) shows that the chemical binding of the hydrogen in monoisopropyldiphenyl is stronger than in water and, accordingly, the neutron thermalization in monoisopropyldiphenyl proceeds at a slower rate.

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The law of scattering $S(\alpha,\beta)$ and the generalized frequency spectrum for water and monoisopropyldiphenyl were obtained from the experimental data.

2. Investigation of the establishment of the thermal neutron spectrum in time in various moderators.

The establishment of the thermal neutron spectrum was studied with the aid of a pulsed neutron source and a mechanical chopper. Neutrons were produced on a target of a linear accelerator which operated in phase with the mechanical chopper. Neutron spectra were measured in beryllium and graphite at different time intervals after fast neutron burst. The resolution during measurements was 3 to 5 μsec/m, the duration of the neutron pulse furnished by the mechanical chopper, 5 to 10 μsec.

3. Investigation of thermal and slowing-down neutron spectra in homogeneous and heterogeneous media.

Microbeams of neutrons were extracted from the medium under investigation. The neutron spectra in the beams were studied with the aid of a mechanical chopper using the time-of-flight method. The resolution during the measurements of the thermal part of the spectrum was 10 to 25 μsec/m, during the measurements of the slowing-down spectrum, 1.2 μsec/m. Spectra of thermal neutrons in boron-poisoned water were studied at various boron concentrations.

A detailed investigation was made into the space-energy distribution of neutrons in the cells of subcritical lattices: uranium-water, uranium-monoisopropyldiphenyl, uranium-graphite.

The measurement results indicate that the thermal neutron spectrum in water can be described by the Maxwell distribution with the temperature of the neutrons exceeding the temperature of the media. At the given temperature of the moderator the temperature rise of the neutrons is proportional to absorption.

The spectrum of thermal neutrons in the uranium block of a lattice and also in a moderator differs from equilibrium Maxwell distribution. The nature of the neutron spectrum abruptly changes at the boundary between uranium and moderator. Inside the block and the moderator the neutron spectrum changes but little. A simple semi-empirical relation is given between the thermal neutron spectrum in the block and in the moderator. The spectrum of the slowing-down neutrons in the block has peculiar dips at energies corresponding to the known absorption levels of U^{238} and U^{235}.

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Abstract

EXPERIMENTAL STUDIES ON FAST - NEUTRON REACTOR PHYSICS

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This report deals with the experimental study of some problems of fast-neutron reactor physics. These studies continue a series of experiments on fast reactor physics that has been carried out at the Physical-Energy Institute since 1950. The first results of this experimental series had been reported at the 2-nd International Conference on the Peaceful Uses of Atomic Energy in 1958.

In this report the following problems are considered:

I. Refinement and development of multigroup cross-section sets for nuclear reactors calculation. The importance of different physical effects influencing neutron propagation in media and the possibility of taking them into account in multigroup calculations are discussed (resonance cross-section self-shielding, anisotropy of angular distributions of scattered neutrons, heterogeneous effects, etc.).

2. Studies of neutron propagation in large blocks of material performed on reactor BR-I are described. The results of neutron space-energy distribution measurements in uranium metal, uranium oxide and carbide, thorium, iron, nickel and copper are given. Experimental data of albedo characteristics for different materials (iron, nickel, water, beryllium, etc.) at the boundary with uranium, thorium and uranium carbide are considered.

These investigations are supplemented with spectral measurements of neutron leaving some media (iron, nickel, stainless steel, uranium) that were performed by means of time-of-flight technique on the reactor IBR with 0,04 sec./m reso-

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lution. The results of all the experiments are compared with multigroup calculation data. Cross-section resonance structure effect is revealed as well as the accuracy of taking into account resonance structure values in calculations.

3. The reactor BFS intended for large fast neutron critical assemblies investigation is described. The results of several such assemblies studies with loadings up to some hundreds kilograms of U^{235} are given.

The experiments conducted permitted to define critical masses, conversion ratios, neutron spectral characteristics in different points of the assemblies studied, reactivity perturbations caused by the introduction of some materials in different points of a system, etc.

The problems of the influence of reactor components heterogeneous arrangement (caused by the reactor BFS design features) on physical parameters and neutron space-energy distributions were studied.

The experimental results are compared with the multigroup calculation data.

4. The results of radiative capture cross-section measurements on fissionable isotopes for neutron spectra of the BR-5 core and reflector are given. The measurements were based on detecting of isotope change in materials subjected to long-term irradiation in the reactor.

The results of measurements are compared with miltigroup calculation data. The accuracy of calculations was checked by comparison with experimental distribution of different reactions with the known energy cross-section dependence. The choice of neutron radiative capture group constants for the isotopes studied is discussed.

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Abstract

CALCULATIONS ON POWER FAST REACTORS PHYSICS

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The main feature of fast reactors is the possibility to develop on their own fuel. The rate of this development is defined both by the characteristics of the reactor itself and by the parameters of fuel cycle: burnup, reprocessing time, fuel losses during reprocessing. Equations describing reactor breeding both for one kind of fuel and for the case of different fuels mixture are formulated. The dependence of fuel doubling time on the characteristics of fast reactor developing system is obtained.

Reactor characteristics are defined primarily by the kind of fuel and by possible heat transfer conditions in the core and blankets. Two different possibilities are considered from this point of view. The first one is connected with the use of oxide fuel and with such heat transfer conditions that are based on design studies and experiments. The second possibility is connected with the future increase of coolant parameters, of its heating and with the use of thin-canned carbide fuel element.

Heat engineering restrictions in powerful reactors do not permit to place sufficient amounts of fuel and fertile material in the core. In this connection the means to decrease the volume fraction of coolant and to increase fuel and fertile material content are considered. One way to achieve this is to flatten the core, and another one is to vary phy-

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sical properties over the core. The significance of power distribution flattening and different means of its realizotion are considered.

Power distribution flattening is shown to allow the improving of reactor physical characteristics.

Both different physical reactor characteristics as a function of composition and the significance of different materials for neutron balance are analyzed.

Reactor breeding rate as a function of reactor characteristics and reprocessing parameters is shown on the basis of doubling time comparison.

Optimum power density (from the doubling time point of view) and its dependence on the reactor power at different original assumptions is considered.

The results of thorium cycle studies for fast reactors are given, joint cycle is proposed where plutonium and uranium-233 are used as fuel and uranium-238 - as fertile material in the core. Thorium serves as fertile material in the blanket. As a result uranium-233 is generated in blankets and plutonium - in the cores.

The principle characteristics of separate and joint systems are analyzed, doubling time comparison is performed and the rate of joint system breeding is shown to be not much less than plutonium system breeding rate.